September 4, 2008

EA-08-250

Dr. Donald Wall, Director
Nuclear Radiation Center
Washington State University
PO Box 641300
Pullman, WA 99164-1300

SUBJECT: ISSUANCE OF ORDER MODIFYING LICENSE NO. R-76 TO CONVERT FROM HIGH- TO LOW-ENRICHED URANIUM FUEL (AMENDMENT NO. 20) – WASHINGTON STATE UNIVERSITY TRIGA REACTOR (TAC NO. MD6570)

Dear Dr. Wall:

The U.S. Nuclear Regulatory Commission (NRC) is issuing the enclosed Order, as Amendment No. 20 to Facility Operating License No. R-76, which authorizes the conversion of the Washington State University TRIGA Reactor from high-enriched uranium fuel to low-enriched uranium (LEU) fuel. This Order modifies the license, including the technical specifications, in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.64. This regulation requires that non-power reactor licensees, such as the Washington State University, convert to LEU fuel under certain conditions which Washington State University now meets. The Order is being issued in accordance with 10 CFR 50.64(c)(3) and in response to your submittal of August 15, 2007, as supplemented on December 14, 2007, and January 15, June 13, and August 4, 22, and 25, 2008. The Order also contains an outline of a reactor startup report that you are required to provide to the NRC within 6 months following the return of the converted reactor to normal operation.

The Order becomes effective on the later date of either the day of receipt of an adequate number and type of LEU fuel elements that are necessary to operate the facility as specified in your submittal and supplements, or 20 days after the date of its publication in the *Federal Register*, provided there are no requests for a hearing.

Although this Order is not subject to the requirements of the Paperwork Reduction Act, there is nonetheless a clearance from the Office of Management and Budget (OMB), OMB approval number 3150-0012, that covers the information collections contained in the Order.

Copies of replacement pages for the facility operating license and technical specifications, and of the NRC staff safety evaluation for the conversion to LEU fuel are also enclosed. The Order is being sent to the *Federal Register* for publication.

Sincerely,

/RA/

Alexander Adams, Jr., Senior Project Manager
Research and Test Reactors Branch A
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Docket No. 50-027

Enclosures:
1. Order
2. Replacement Pages for License
3. Replacement Pages for Technical Specifications
4. Safety Evaluation

cc w/enclosures: See next page

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cc w/enclosures: See next page

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NAME	EBarnhill eeb	AAdams aa	SUttal: NLO w/changes su	NHilton nh
DATE	9/3/08	8/29/08	9/2/08	9/12/08
OFFICE	PRTA/BC	DPR/D	NRR/D	PRTA/PM
NAME	DCollins dsc	MCase mc	JWiggins for ELeeds	AAdams aa
DATE	9/3/08	9/3/08	9/4/08	9/4/08

OFFICIAL RECORD COPY

cc:

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Chair, Reactor Safeguards Committee
Nuclear Radiation Center
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Office of the Governor
Executive Policy Division
State Liaisons Officer
P.O. Box 43113
Olympia, WA 98504-3113

Test, Research, and Training
Reactor Newsletter
University of Florida
202 Nuclear Sciences Center
Gainesville, FL 32611

7590-01-P

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
WASHINGTON STATE UNIVERSITY	)	Docket No. 50-027
	)	EA-08-250
(Washington State University TRIGA Reactor)	)	

ORDER MODIFYING FACILITY OPERATING LICENSE NO. R-76

I.

Washington State University (the licensee) is the holder of Amended Facility Operating License No. R-76 (the license) originally issued on March 6, 1961, by the U.S. Atomic Energy Commission and subsequently renewed on August 11, 1982, by the U.S. Nuclear Regulatory Commission (the NRC or the Commission). The license authorizes operation of the Washington State University TRIGA Reactor (the facility) at a power level up to 1,000 kilowatts thermal and to receive, possess, and use special nuclear material associated with the operation. The facility is a research reactor located on the campus of the Washington State University, in the city of Pullman, Whitman County, Washington. The mailing address is Nuclear Radiation Center, Washington State University, P.O. Box 641300, Pullman, Washington 99164-1300.

II.

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.64, limits the use of high-enriched uranium (HEU) fuel in domestic non-power reactors (research and test reactors) (see 51 FR 6514). The regulation, which became effective on March 27, 1986, requires that if Federal Government funding for conversion-related costs is available, each licensee of a non-power reactor authorized to use HEU fuel shall replace it with low-enriched uranium (LEU) fuel acceptable to the Commission unless the Commission has determined that the reactor has a unique purpose. The Commission's stated purpose for these requirements was to reduce, to

the maximum extent possible, the use of HEU fuel in order to reduce the risk of theft and diversion of HEU fuel used in non-power reactors.

Paragraphs 50.64(b)(2)(i) and (ii) require that a licensee of a non-power reactor (1) not acquire more HEU fuel if LEU fuel that is acceptable to the Commission for that reactor is available when the licensee proposes to acquire HEU fuel, and (2) replace all HEU fuel in its possession with available LEU fuel acceptable to the Commission for that reactor in accordance with a schedule determined pursuant to 10 CFR 50.64(c)(2).

Paragraph 50.64(c)(2)(i) requires, among other things, that each licensee of a non-power reactor authorized to possess and to use HEU fuel develop and submit to the Director of the Office of Nuclear Reactor Regulation (the Director) by March 27, 1987, and at 12-month intervals thereafter, a written proposal for meeting the requirements of the rule. The licensee shall include in its proposal a certification that Federal Government funding for conversion is available through the U.S. Department of Energy or other appropriate Federal agency. The proposal should also provide a schedule for conversion, based upon availability of replacement fuel acceptable to the Commission for that reactor and upon consideration of other factors such as the availability of shipping casks, implementation of arrangements for available financial support, and reactor usage.

Paragraph 50.64(c)(2)(iii) requires the licensee to include in the proposal, to the extent required to effect conversion, all necessary changes to the license, the facility, and licensee procedures. This paragraph also requires the licensee to submit supporting safety analyses in time to meet the conversion schedule.

Paragraph 50.64(c)(2)(iii) also requires the Director to review the licensee proposal, to confirm the status of Federal Government funding, and to determine a final schedule, if the licensee has submitted a schedule for conversion.

Section 50.64(c)(3) requires the Director to review the supporting safety analyses and to issue an appropriate enforcement order directing both the conversion and, to the extent

consistent with the protection of public health and safety, any necessary changes to the license, the facility, and licensee procedures. In the *Federal Register* notice of the final rule (51 FR 6514), the Commission explained that in most, if not all cases, the enforcement order would be an order to modify the license under 10 CFR 2.204 (now 10 CFR 2.202).

Any person, other than the licensee, whose interest may be affected by this proceeding and who desires to participate as a party must file a written request for hearing or petition for leave to intervene meeting the requirements of 10 CFR 2.309, “Hearing Requests, Petitions to Intervene, Requirements for Standing, and Contentions.”

III.

The U.S. Nuclear Regulatory Commission (NRC) maintains the Agencywide Documents Access and Management System (ADAMS), which provides text and image files of the NRC's public documents. On August 15, 2007 (ADAMS Accession Nos. ML072410493 and ML080170058), as supplemented on December 14, 2007 (ADAMS Accession No. ML080090628), and January 15 (ADAMS Accession No. ML080170037), June 13, (ADAMS package Accession No. ML082380270 which consists of ADAMS Accession Nos. ML082380265, ML082380266, ML082380267, ML082380268, ML082380269, ML082380271, ML08238272, ML082380273 and ML082380279) and August 4, (ADAMS Accession No.ML082210118), 22, (ADAMS Accession No. ML082390030), and 25, 2008 (ADAMS Accession No. ML082400522), the NRC staff received the licensee's conversion proposal, including its proposed modifications and supporting safety analyses. HEU fuel elements are to be replaced with LEU fuel elements. The reactor core contains fuel clusters, each fuel cluster contains up to four fuel elements of the TRIGA design, with the fuel consisting of uranium-zirconium hydride with 30 weight percent uranium. These fuel elements contain the uranium-235 isotope at an enrichment of less than 20 percent. The NRC staff reviewed the licensee's proposal and the requirements of 10 CFR 50.64 and has determined that public health and safety and common defense and security require the licensee to convert the facility from the use

of HEU to LEU fuel in accordance with the attachments to this Order and the schedule included herein. The attachments to this Order specify the changes to the license conditions and technical specifications that are needed to amend the facility license and contains an outline of a reactor startup report to be submitted to NRC within 6 months following return of the converted reactor to normal operation.

IV.

Accordingly, pursuant to Sections 51, 53, 57, 101, 104, 161b, 161i, and 161o of the Atomic Energy Act of 1954, as amended, and to Commission regulations in 10 CFR 2.202 and 10 CFR 50.64, IT IS HEREBY ORDERED THAT:

Amended Facility Operating License No. R-76 is modified by amending the license conditions and technical specifications as stated in the attachments to this Order (Attachment 1: MODIFICATIONS TO FACILITY OPERATING LICENSE NO. R-76; Attachment 2: OUTLINE OF REACTOR STARTUP REPORT). The Order becomes effective on the later date of either (1) the day the licensee receives an adequate number and type of LEU fuel elements to operate the facility as specified in the licensee proposal dated August 15, 2007 (ADAMS Accession Nos. ML072410493 and ML080170058), as supplemented on December 14, 2007 (ADAMS Accession No. ML080090628), and January 15, (ADAMS Accession No. ML080170037), June 13, (ADAMS package Accession No. ML082380270 which consists of ADAMS Accession Nos. ML082380265, ML082380266, ML082380267, ML082380268, ML082380269, ML082380271, ML08238272, ML082380273 and ML082380279) and August 4, (ADAMS Accession No.ML082210118), 22, (ADAMS Accession No. ML082390030), and 25, 2008 (ADAMS Accession No. ML082400522), or (2) 20 days after the date of publication of this Order in the Federal Register.

V.

Pursuant to 10 CFR 2.202, any person(s) whose interest may be affected by this proceeding, other than the licensee, and who wishes to participate as a party in the proceeding

must file a written request within 20 days after the date of publication of this Order, setting forth with particularity the manner in which this Order adversely affects his or her interest and addressing the criteria set forth in 10 CFR 2.309. If a hearing is held, the issue to be considered at such hearing shall be whether this Order should be sustained.

A request for a hearing must be filed in accordance with the NRC E-Filing rule, which became effective on October 15, 2007. The NRC issued the E-filing final rule on August 28, 2007, (72 FR 49139) and codified it in pertinent part at 10 CFR Part 2, "Rules of Practice for Domestic Licensing Proceedings and Issuance of Orders," Subpart B. The E-Filing process requires participants to submit and serve documents over the Internet or, in some cases, to mail copies on electronic optical storage media. Participants may not submit paper copies of their filings unless they seek a waiver in accordance with the procedures described below.

To comply with the procedural requirements associated with E-Filing, at least 10 days before to the filing deadline, the requestor must contact the Office of the Secretary by email at hearingdocket@nrc.gov, or by calling (301) 415-1677, to request (1) a digital identification (ID) certificate, which allows the participant (or its counsel or representative) to digitally sign documents and access the E-Submittal server for any NRC proceeding in which it is participating, and/or (2) creation of an electronic docket for the proceeding (even in instances when the requestor (or its counsel or representative) already holds an NRC-issued digital ID certificate). Each requestor will need to download the Workplace Forms Viewer™ to access the Electronic Information Exchange (EIE), a component of the E-Filing system. The Workplace Forms Viewer™ is free and is available at http://www.nrc.gov/site-help/e-submittals/install-viewer.html. Information about applying for a digital ID certificate also is available on the NRC's public Web site at http://www.nrc.gov/site-help/e-submittals/apply-certificates.html.

Once a requestor has obtained a digital ID certificate, had a docket created, and downloaded the EIE viewer, he or she can then submit a request for a hearing through EIE. Submissions should be in portable document format (PDF) in accordance with NRC guidance

available on the NRC public Web site at http://www.nrc.gov/site-help/e-submittals.html. A filing is considered complete at the time the filer submits its document through EIE. To be timely, electronic filings must be submitted to the EIE system no later than 11:59 p.m. eastern time on the due date. Upon receipt of a transmission, the E-Filing system time-stamps the document and sends the submitter an email notice confirming receipt of the document. The EIE system also distributes an email notice that provides access to the document to the NRC Office of the General Counsel and any others who have advised the Office of the Secretary that they wish to participate in the proceeding, so that the filer need not serve the document on those participants separately. Therefore, any others who wish to participate in the proceeding (or their counsel or representative) must apply for and receive a digital ID certificate before a hearing request is filed so that they may obtain access to the document via the E-Filing system.

A person filing electronically may seek assistance through the "Contact Us" link located on the NRC Web site at http://www.nrc.gov/site-help/e-submittals.html or by calling the NRC technical help line, which is available between 8:30 a.m. and 4:15 p.m., eastern time, Monday through Friday. The help line number is (800) 397-4209 or, locally, (301) 415-4737.

Participants who believe that they have good cause for not submitting documents electronically must file a motion, in accordance with 10 CFR 2.302(g), with their initial paper filing requesting authorization to continue to submit documents in paper format. Such filings must be submitted by (1) first class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Rulemaking and Adjudications Staff; or (2) courier, express mail, or expedited delivery service to the Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville, Pike, Rockville, Maryland, 20852, Attention: Rulemaking and Adjudications Staff. Participants filing a document in this manner are responsible for serving the document on all other participants. Filing is considered complete by first-class mail as of the time of deposit in the mail, or by

courier, express mail, or expedited delivery service upon depositing the document with the provider of the service.

Documents submitted in adjudicatory proceedings will appear in the NRC's electronic hearing docket at http://ehd.nrc.gov/EHD_Proceeding/home.asp, unless excluded pursuant to an order of the Commission, an Atomic Safety and Licensing Board, or a Presiding Officer. Participants are requested not to include personal privacy information, such as social security numbers, home addresses, or home phone numbers, in their filings. With respect to copyrighted works, except for limited excerpts that serve the purpose of the adjudicatory filings and would constitute a fair use application, participants are requested not to include copyrighted materials in their works.

If a hearing is requested and granted by the Commission, the NRC will issue an order designating the time and place of any hearing.

In the absence of any request for hearing, the provisions as specified in Section IV shall be final twenty (20) days after the date of publication of this Order in the *Federal Register*.

In accordance with 10 CFR 51.10(d) this Order is not subject to Section 102(2) of the National Environmental Policy Act, as amended. The NRC staff notes, however, that with respect to environmental impacts associated with the changes imposed by this Order as described in the safety evaluation, the changes would, if imposed by other than an order, meet the definition of a categorical exclusion in accordance with 10 CFR 51.22(c)(9). Thus, pursuant to either 10 CFR 51.10(d) or 51.22(c)(9), no environmental assessment or environmental impact statement is required.

Detailed guidance which the NRC uses to review applications from research reactor licensees appears in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," February 1996, which can be obtained from the Commission's Public Document Room (PDR). The public may also access NUREG-1537 through the NRC's Public Electronic Reading Room on the Internet at

http://www.nrc.gov/reading-rm/adams.html under ADAMS Accession Nos. ML0412430055 for part one and ML042430048 for part two.

For further information see the application from the licensee dated August 15, 2007 (ADAMS Accession Nos. ML072410493 and ML080170058), as supplemented on December 14, 2007 (ADAMS Accession No. ML080090628), and January 15, (ADAMS Accession No. ML080170037), June 13, (ADAMS package Accession No. ML082380270 which consists of ADAMS Accession Nos. ML082380265, ML082380266, ML082380267, ML082380268, ML082380269, ML082380271, ML08238272, ML082380273 and ML082380279) and August 4, (ADAMS Accession No.ML082210118), 22, (ADAMS Accession No. ML082390030), and 25, 2008 (ADAMS Accession No. ML082400522), the NRC staff's requests for additional information (ADAMS Accession Nos. ML073240018, ML080460523 and ML082250618), and the cover letter to the licensee and the staff's safety evaluation dated September 4, 2008 (ADAMS Accession No. ML082401484). On January 23, 2008, the NRC staff issued an Order to the licensee to allow receipt and possession of the special nuclear material needed for the conversion (ADAMS Accession No. ML073550839). These documents are available for public inspection at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area O1 F21, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible electronically from the ADAMS Public Electronic Reading Room on the Internet at the NRC Web site, http://www.nrc.gov/reading-rm/adams.html.

Persons who do not have access to ADAMS or who have problems in accessing the documents in ADAMS should contact the NRC PDR reference staff by telephone at 1-800-397-4209 or 301-415-4737 or by e-mail to pdr@nrc.gov.

Dated this 4th day of September 2008.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

James T. Wiggins, Deputy Director
Office of Nuclear Reactor Regulation

Attachments:
1. Modifications to Facility Operating License No. R-76
2. Outline of Reactor Startup Report

MODIFICATIONS TO FACILITY OPERATING LICENSE NO. R-76

A. License Conditions Revised by This Order

2.B.(2) Pursuant to the Act and 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material”, to receive, possess, and use in connection with operation of the reactor:

a. up to 25 kilograms of contained uranium-235 enriched to less than 20 percent in the form of TRIGA reactor fuel.

b. up to 500 grams of contained uranium-235 enriched to any enrichment in the form of nuclear detectors and material for experimental research.

c. up to 32 grams of plutonium in the form of a plutonium-beryllium neutron source.

2.B.(4) Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," to possess, but not use, up to 15 kilograms of contained uranium-235 at equal to or greater than 20 percent enrichment in the form of TRIGA fuel until the existing inventory of this fuel is removed from the facility.

2.C.(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 20 are, hereby, incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

OUTLINE OF REACTOR STARTUP REPORT

Within six months following the return of the converted reactor to normal operation, submit the following information to the NRC. Information on the HEU core should be presented to the extent it exists.

1. Critical mass

   Measurement with HEU
   Measurement with LEU
   Comparisons with calculations for LEU and if available, HEU

2. Excess (operational) reactivity

   Measurement with HEU
   Measurement with LEU
   Comparisons with calculations for LEU and if available, HEU

3. Regulating and safety control rod calibrations

   Measurement of HEU and LEU rod worths and comparisons with calculations for LEU and if available, HEU

4. Reactor power calibration

   Methods and measurements that ensure operation within the license limit and comparison between HEU and LEU nuclear instrumentation set points, detector positions and detector output.

5. Shutdown margin

   Measurement with HEU
   Measurement with LEU
   Comparisons with calculations for LEU and if available, HEU

6. Thermal neutron flux distributions

   Measurements of the core and measured experimental facilities (to the extent available) with HEU and LEU and comparisons with calculations for LEU and if available, HEU.

7. Reactor physics measurements

   Results of determination of LEU effective delayed neutron fraction, temperature coefficient, and void coefficient to the extent that measurements are possible and comparison with calculations and available HEU core measurements.

8. Initial LEU core loading

Measurements made during initial loading of the LEU fuel, presenting subcritical multiplication measurements, predictions of multiplication for next fuel additions, and prediction and verification of final criticality conditions.

9. Primary coolant measurements

Results of any primary coolant water sample measurements for fission product activity taken during the first 30 days of LEU operation.

10 Pulse Measurements

Results of any test pulses performed and comparison with calculations and available HEU core measurements

11. Discussion of results

Discussion of the comparison of the various results including an explanation of any significant differences that could affect normal operation and accident analyses.

FACILITY OPERATING LICENSE NO. R-76

DOCKET NO. 50-027

REPLACEMENT PAGE FOR LICENSE

Replace the following pages of the License with the enclosed pages. The revised pages are identified by amendment number and contains a vertical line indicating the area of change.

Remove	Insert
3	3
4	4

(2) Pursuant to the Act and 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material”, to receive, possess, and use in connection with operation of the reactor:

Amendment 20

a. up to 25 kilograms of contained uranium-235 enriched to less than 20 percent in the form of TRIGA reactor fuel.

b. up to 500 grams of contained uranium-235 enriched to any enrichment in the form of nuclear detectors and material for experimental research.

c. up to 32 grams of plutonium in the form of a plutonium-beryllium neutron source.

(3) Pursuant to the Act and 10 CFR Part 30, “Rules of General Applicability to Licensing of Byproduct Material,” and Part 70, to possess, but not separate, such byproduct and special nuclear materials as may have been produced and may be produced by the operation of the facility.

(4) Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," to possess, but not use, up to 15 kilograms of contained uranium-235 at equal to or greater than 20 percent enrichment in the form of TRIGA fuel until the existing inventory of this fuel is removed from the facility.

Amendment 20

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I; Part 20, Section 30.34 of Part 30, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70, is subject to all applicable provisions of the Act and to the rules, and orders of the Commission now, or hereafter, in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

The licensee is authorized to operate the facility at steady state power levels not in excess of 1000 kilowatts (thermal) and to pulse the reactor in accordance with the limitations in the Technical Specifications.

Amendment 12 1/23/1990

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 20, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Physical Security Plan

The licensee shall maintain and fully implement all provisions of the Commission-approved physical security plan, in accordance with 10 CFR 73.67(d) and (e), including amendments and changes made pursuant to the authority of 10 CFR 50.54(p). The approved physical security plan consists of documents withheld from public disclosure pursuant to 10 CFR 73.21 entitled "Physical Security Plan for Washington State University," dated November 9, 1983, submitted by letter dated November 11, 1983, as amended by letter dated July 18, 1984.

D. This amended license is effective as of the date of issuance and shall expire at midnight twenty years from date of issuance.

R/A

Harold Bernard, Acting Branch Chief
Standardization & Special
Projects Branch
Division of Licensing

Enclosure:
Appendix A –
Technical Specifications, July 1982

Date of Issuance: August 11, 1982

FACILITY OPERATING LICENSE NO. R-76

DOCKET NO. 50-027

REPLACEMENT PAGES FOR TECHNICAL SPECIFICATIONS

Replace the following pages of Appendix A, Technical Specifications, with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
3	3
4	4
6	6
7	7
8	8
	8A
9	9
10	10
30	30
	30A
31	31
34	34

(6) release of fission products into the environment

Shutdown Margin: Shutdown margin shall mean the minimum shutdown reactivity necessary to provide confidence that (1) the reactor can be made subcritical by means of the control and safety systems, starting from any permissible operating conditions, and (2) the reactor will remain subcritical without further operator action.

Steady-State Mode: Steady-state mode operation shall mean any operation of the reactor with the mode selector switch in the steady-state position.

1.2 Reactor Experiments and Irradiations

Experiment: Experiment shall mean: (1) any apparatus, device or material which is not a normal part of the core or experimental facilities, but which is inserted into these facilities or is in line with a beam of radiation originating from the reactor, or (2) any operation designed to measure reactor parameters or characteristics.

Experimental Facilities: Experimental facilities shall mean beam ports, including extension tubes with shields, thermal columns with shields, vertical tubes, in-core irradiation baskets or tubes, pneumatic transfer systems, and any other in-pool irradiation facilities.

Irradiation: Irradiation shall mean the insertion of any device or material that is not a normal part of the core or experimental facilities into an irradiation facility so that the device or material is exposed to a significant amount of the radiation available in that irradiation facility.

Irradiation Facilities: Any in-pool experimental facility that is not a normal part of the core and that is used to irradiate devices and materials.

Secured Experiment: A secured experiment shall mean any experiment that is held firmly in place by a mechanical device or by gravity, that is not readily removable from the reactor, and that requires one of the following actions to permit removal:

(1) removal of mechanical fasteners
(2) use of underwater handling tools
(3) moving of shield blocks or beam port components

1.3 Reactor Component

30/20 LEU Fuel: 30/20 LEU fuel is TRIGA fuel that contains a nominal 30 weight percent of uranium with a nominal ^{235}U enrichment of 19.75% and erbium, a burnable poison.

Fuel Bundle: A fuel bundle is a cluster of three or four fuel rods fastened together in a square array by a top handle and bottom grid plate adapter.

Fuel Rod: A fuel rod is a single TRIGA-type fuel rod of either Standard or 30/20 LEU fuel.

Instrumented Fuel Rod: An instrumented fuel rod is a special fuel rod in which thermocouples have been embedded for the purpose of measuring the fuel temperatures during reactor operation.

Mixed Core: A mixed core is a core arrangement containing Standard and 30/20 LEU-type fuels with at least 51 30/20 LEU fuel rods located in the central positions in the core.

Operational Core: An operational core is any arrangement of TRIGA fuel that is capable of operating at the maximum licensed power level and that satisfies all the requirements of the Technical Specifications.

Regulating Control Element: Regulating control element shall mean a low worth control element that may be positioned either manually or automatically by means of an electric motor-operated positioning system and that need not have a scram capability.

Standard Control Element: Standard control element shall mean any control element that has a scram capability, that is utilized to vary the reactivity of the core, and that is positioned by means of an electric motor-operated positioning system.

Standard Core: A standard core is any arrangement of all-Standard fuel.

Standard Fuel: Standard fuel is TRIGA fuel that contains a nominal 8.5 weight percent of uranium with a ^{235}U enrichment of less than 20%.

Transient Control Element: Transient control element shall mean any control element that has the capability of being rapidly withdrawn from the reactor core by means of a pneumatic drive, that is capable of being positioned by means of an electric motor-operated positioning system, and that has scram capabilities.

1.4 Reactor Instrumentation

Channel Calibration: A channel calibration consists of comparing a measured value from the measuring channel with a corresponding known value of the parameter so that the measuring channel output can be adjusted to respond with acceptable accuracy to known values of the measured variables.

Channel Check: A channel check is a qualitative verification of acceptable performance by observation of channel behavior. This verification may include comparison with independent channels measuring the same variable or other measurements of the variables.

Channel Test: A channel test is the introduction of a signal into the channel to verify that it is operable.

Experiment Safety Systems: Experiment safety systems are those systems, including their associated input circuits, that are designed to initiate a scram for the primary purpose of protecting an experiment or to provide information that requires manual protective action to be initiated.

Limiting Safety Systems Setting: Limiting safety systems settings are the settings for automatic protective devices related to those variables having significant safety functions.

Measured Value: The measured value is the magnitude of that variable as it appears on the output of a measuring channel.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limit - Fuel Element Temperature

Applicability: This specification applies to the temperature of the reactor fuel.

Objective: The objective is to define the maximum fuel temperature that can be permitted with confidence that a fuel cladding failure will not occur.

Specifications:

(1) The maximum temperature in a Standard TRIGA fuel rod shall not exceed 1000°C under any condition of operation.

(2) The maximum temperature in a 30/20 LEU-type TRIGA fuel rod shall not exceed 1150°C under any condition of operation.

Bases: The important parameter for a TRIGA reactor is the fuel rod temperature. This parameter is well-suited as a single specification, especially since it can be measured. A loss in the integrity of the fuel rod cladding could arise from a buildup of excessive pressure between the fuel moderator and the cladding if the fuel temperature exceeds the safety limit. The pressure is caused by the presence of air, fission product gases, and hydrogen from the disassociation of the hydrogen and zirconium in the fuel moderator. The magnitude of this pressure is determined by the fuel-moderator temperature and the ratio of hydrogen to zirconium in the alloy. The safety limit for the 30/20 LEU fuel is based on data that indicate that the stress in the cladding because of the hydrogen pressure from the disassociation of zirconium hydride will remain below the ultimate stress, provided the temperature of the fuel does not exceed 1150°C and the fuel cladding is water cooled.* The safety limit for the Standard TRIGA fuel is based on data, including the large mass of experimental evidence obtained during high performance reactor tests on this fuel. These data indicate that the stress in the cladding because of hydrogen pressure from the disassociation of zirconium hydride will remain below the ultimate stress, provided that the temperature of the fuel does not exceed 1000°C and the fuel cladding is water cooled.*

2.2 Limiting Safety System Settings

Applicability: This specification applies to the settings that prevent the safety limit from being reached.

Objective: The objective is to prevent the safety limits from being reached.

Specifications: The limiting safety system settings shall be 500˚C as measured in an instrumented fuel rod in selected locations in the central region of the core. For a mixed core, the instrumented rod shall be located in one of the following positions in the region of the core containing the 30/20 LEU-type fuel rods: D2NE, D2SE, C3 (except for C3NE), D3, E3 (except for E3SE), C4, E4NE, E4NW, C5 (except for C5SW), D5SE, E5NE, E5NW, C6NW, or D6.

*GA-9064, Safety Analysis Report for the Torrey Pines TRIGA Mark III Reactor, submitted under Docket No. 50-227.

Bases: The limiting safety system setting is the measured instrumented fuel rod temperature that, if exceeded, shall initiate a scram to prevent the fuel temperature safety limit from being exceeded. The response to Question number 40 of the additional RAI for the FLIP-to-30/20 LEU conversion safety analysis report for the Washington State University (WSU) TRIGA reactor showed that for both the hottest and the coldest thermocouples, an IFE located in core positions D2NE, D2SE, C3 (except for C3NE), D3, E3 (except for E3SE), C4, E4NE, E4NW, C5 (except for C5SW), D5SE, E5NE, E5NW, C6NW, or D6 would protect the fuel temperature safety limit of 1150°C for reactor power levels that are less than 1.7 MW and limit the maximum steady-state temperature in the 30/20 fuel region to less than 800°C. This setting provides at least a 350°C margin of safety for 30/20 fuel and at least a 200°C margin of safety for Standard fuel.

In the pulse mode of operation, the same limiting safety system setting will apply. However, the temperature channel will not limit the peak power generated during the pulse because of the relatively long response time of the temperature channel as compared with the width of a pulse. On the other hand, the temperature scram would limit the total amount of energy generated in a pulse by cutting off the "tail" of the energy transient in the event that the fuel temperature limit is exceeded. Thus, the fuel temperature scram provides an additional degree of safety in the pulse mode of operation to protect the fuel in the event of such conditions as sticking of the transient control element in the withdrawn position after a pulse.

3.0 LIMITING CONDITIONS OF OPERATION

3.1 Steady-State Operation

Applicability: This specification applies to the energy generated in the reactor during steady-state operation.

Objective: The objective is to ensure that the fuel temperature safety limit will not be exceeded during steady-state operation.

Specifications: The reactor power level shall not exceed 1.3 MW under any condition of operation. The normal steady-state operating power level of the reactor shall be 1.0 MW. However, for purposes of testing and calibration, the reactor may be operated at higher power levels not to exceed 1.3 MW during the testing period.

Basis: Thermal and hydraulic calculations performed by the vendor indicate that TRIGA fuel may be safely operated up to power levels of at least 2.0 MW with natural convection cooling.

3.2 Reactivity Limitations

Applicability: These specifications apply to the reactivity condition of the reactor and the reactivity worth of control elements and experiments. They apply for all modes of operation.

Objective: The objective is to ensure that the reactor can be shut down at all times and to ensure that the fuel temperature safety limit will not be exceeded.

Specifications: The reactor shall not be operated unless the shutdown margin provided by control elements shall be 0.25$ or greater with:

(1) the highest worth nonsecured experiment in its most reactive state

(2) the highest worth control element and the regulating element (if not scrammable) fully withdrawn

(3) the reactor in the cold critical condition without xenon

Basis: The value of the shutdown margin ensures that the reactor can be shut down from any operating condition even if the highest worth rod should remain in the fully withdrawn position. If the regulating rod is not scrammable, its worth is not used in determining the shutdown reactivity.

3.3 Pulse Mode Operation

Applicability: This specification applies to the peak fuel temperature in the reactor as a result of a pulse insertion of reactivity.

Objective: The objective is to ensure that fuel element damage does not occur in any fuel rod during pulsing.

Specifications: The maximum reactivity inserted during pulse mode operation shall be such that the peak fuel temperature in any fuel rod in the core does not exceed 830°C. The maximum

safe allowable reactivity insertion shall be calculated annually for an existing core and prior to pulsing a new or modified core arrangement.

Basis: TRIGA fuel is fabricated with a nominal hydrogen to zirconium ratio of 1.6 for 30/20 LEU fuel and 1.65 for Standard. This yields delta phase zirconium hydride which has a high creep strength and undergoes no phase changes at temperatures over 1000°C. However, after extensive steady-state operation at 1 MW, the hydrogen will redistribute due to migration from the central high temperature regions of the fuel to the cooler outer regions. When the fuel is pulsed, the instantaneous temperature distribution is such that the highest values occur at the surface of the element and the lowest values occur at the center. The higher temperatures in the outer regions occur in fuel with a hydrogen to zirconium ratio that has now substantially increased above the nominal value. This produces hydrogen gas pressures considerably in excess of that expected for $ZrH_{1.6}$. If the pulse insertion is such that the temperature of the fuel exceeds 874°C, then the pressure will be sufficient to cause expansion of microscopic holes in the fuel that grow larger with each pulse. The expansion of the fuel stresses and distorts the fuel rod material which, in turn, can cause overall swelling and distortion of the cladding and entire fuel rod. The pulsing limit of 830°C is obtained by examining the equilibrium hydrogen pressure of zirconium hydride as a function of temperature. The decrease in temperature from 874°C to 830°C reduces hydrogen pressure by a factor of two, which provides an acceptable safety factor. This phenomenon does not alter the steady-state safety limit since the total hydrogen in a fuel element does not change. Thus, the pressure exerted on the clad will not be significantly affected by the distribution of hydrogen within the element.

3.4 Maximum Excess Reactivity

Applicability: This specification applies to the maximum excess reactivity, above cold critical, which may be loaded into the reactor core at any time.

Objective: The objective is to ensure that the core analyzed in the safety analysis report approximates the operational core within reasonable limits.

Specifications: The maximum reactivity in excess of cold, xenon-free critical shall not exceed 5.6% $\Delta k/k$.

Basis: Although maintaining a minimum shutdown margin at all times ensures that the reactor can be shut down, that specification does not address the total reactivity available within the core. This specification, although over-constraining the reactor system, helps ensure that the licensee's operational power densities, fuel temperatures, and temperature peaks are maintained within the evaluated safety limits. The specified excess reactivity allows for power coefficients of reactivity, xenon poisoning, most experiments, and operational flexibility.

3.5 Core Configuration Limitation

Applicability: This specification applies to mixed cores of 30/20 LEU and Standard types of fuel.

Objective: The objective is to ensure that the fuel temperature safety limit will not be exceeded as a result of power peaking effects in a mixed core.

Specifications:

(1) The 30/20-fueled region in a mixed core shall contain at least 51 30/20 fuel rods in a contiguous block of fuel in the central region of the reactor core. Water holes in the 30/20 region shall be limited to nonadjacent single-rod holes.

(2) Each new mixed core configuration shall be evaluated to determine the allowed locations for the IFE (Reference: Response to RAI Question Number 40).

Bases: The limitation on the allowable core configuration of the 30/20 fuel limits power peaking effects. The limitation on power peaking effects ensures that the fuel temperature limit will not be exceeded in a mixed core.

A 500°C safety system setting and the allowed locations for the IFE limit the peak fuel temperature to less than 800°C (Reference: Response to RAI Question Number 40).

3.6 Control and Safety System

3.6.1 Scram Time

Applicability: This specification applies to the time required for the scrammable control rods to be fully inserted from the instant that a safety channel variable reaches the safety system setting.

Objective: The objective is to achieve prompt shutdown of the reactor to prevent fuel damage.

Specifications: The scram time from the instant that a safety system setting is exceeded to the instant that the slowest scrammable control rod reaches its fully inserted position shall not exceed 2 seconds. For purposes of this section, the above specification shall be considered to be satisfied when the sum of the response time of the slowest responding safety channel, plus the fall time of the slowest scrammable control rod, is less than or equal to 2 seconds.

Basis: This specification ensures that the reactor will be promptly shut down when a scram signal is initiated. Experience and analysis have indicated that for the range of transients anticipated for a TRIGA reactor, the specified scram time is adequate to ensure the safety of the reactor.

3.6.2. Reactor Control System

Applicability: This specification applies to the information that must be available to the reactor operator during reactor operation.

Objective: The objective is to require that sufficient information is available to the operator to ensure safe operation of the reactor.

Specifications: The reactor shall not be operated in the specified mode of operation unless the measuring channels listed in Table 3.1 are operable.

5.0 DESIGN FEATURES

5.1 Reactor Fuel

Applicability: This specification applies to the fuel elements used in the reactor core.

Objective: The objective is to ensure that the fuel elements are of such a design and fabricated in such a manner as to permit their use with a high degree of reliability with respect to their physical and nuclear characteristics.

Specifications:

(1) 30/20 LEU Fuel – The individual unirradiated 30/20 fuel elements shall have the following characteristics:

- uranium content: maximum of 30 wt% enriched to a maximum of 19.95% with nominal enrichment of 19.75% U-235.
- hydrogen-to-zirconium ratio (in the ZrH_x):nominal 1.6 H atoms to 1.0 Zr atoms with a maximum H to Zr ratio of 1.65
- natural erbium content (homogeneously distributed): nominal 0.90 wt%.
- cladding: 304 stainless steel, nominal 0.020 in. thick

(2) Standard TRIGA Fuel - The individual unirradiated Standard TRIGA fuel elements shall have the following characteristics:

- uranium content: maximum of 9.0 wt% enriched to less than 20% ^{235}U
- hydrogen-to-zirconium atom ratio (in the ZrH_x): between 1.5 and 1.8
- cladding: 304 stainless steel, nominal 0.020 in. thick

Basis: The fuel specification permits a maximum uranium enrichment of 19.95% in the 30/20 LEU fuel. This is about 1% greater than the design value for 19.75% enrichment. Such an increase in loading would result in an increase in power density of less than 1%. An increase in local power density of 1% reduces the safety margin by less than 2%.

The fuel specification for a single fuel element permits a minimum erbium content of about 5.6% less than the design value of 0.90 wt%. (However, the quantity of erbium in the full core must not deviate from the design value by more than -3.3%). This variation for a single fuel element would result in an increase in fuel element power density of about 1-2%. Such a small increase in local power density would reduce the safety margin by less than 2%.

The maximum hydrogen-to-zirconium ratio of 1.65 could result in a maximum stress under accident conditions in the fuel element clad about a factor of two greater than for a hydrogen-to-zirconium ration of 1.6. This increase in the clad stress during an accident would not exceed the rupture strength of the clad.

A maximum uranium content of 9 wt% for the standard TRIGA elements is about 6% greater than the design value of 8.5 wt%. Such an increase in loading would result in an increase in power density of 6% and reduces the safety margin by at most 10%. The maximum hydrogen-to-zirconium ratio of 1.8 could result in a maximum stress under accident conditions in the fuel element clad about a factor of 2 greater than the value resulting from a hydrogen-to-zirconium

ratio of 1.60. However, this increase in the clad stress during an accident would not exceed the rupture strength of the clad.

5.2 Reactor Core

Applicability: This specification applies to the configuration of fuel and in-core experiments.

Objective: The objective is to ensure that provisions are made to restrict the arrangement of fuel elements and experiments so as to provide assurance that excessive power densities will not be produced.

Specifications:

(1) The core shall be an arrangement of TRIGA uranium-zirconium-hydride fuel-moderator bundles positioned in the reactor grid plate.

(2) The TRIGA core assembly may be composed of Standard fuel, 30/20 LEU fuel, or a combination thereof (mixed cores) provided that the 30/20 LEU fuel region contains at least 51 30/20 LEU fuel rods located in a contiguous block in the central region of the core.

(3) The reactor fueled with a mixture of fuel types shall not be operated with a core lattice position vacant in the 30/20 LEU fuel region. Water holes in the 30/20 LEU region shall be limited to single-rod holes. Vacant lattice positions in the core fuel region shall be occupied with fixtures that will prevent the installation of a fuel bundle.

(4) The reflector, excluding experiments and experimental facilities, shall be water or a combination of graphite, aluminum and water.

Basis: Standard TRIGA cores have been used for years and their characteristics are well-documented. Cores of 30/20 fuel have been tested by General Atomics Co. Calculations of a mixed core (Standard and 30/20) in the WS reactor has shown that such a core may be safety operated.

In mixed cores, it is necessary to arrange 30/20 LEU elements in a contiguous, central region of the core to control flux peaking and power generation peak values in individual elements.

Vacant core lattice positions in the Standard fuel region will contain experiments or an experimental facility to prevent accidental fuel additions to the reactor core. Vacant core positions are not permitted in the 30/20 LEU fuel region as specified by Section 3.5.

The core will be assembled in the reactor grid plate which is located in a pool of light water. Water in combination with graphite reflectors can be used for neutron economy and the enhancement of experimental facility radiation requirements.

5.3 Control Elements

Applicability: This specification applies to the control elements used in the reactor core.

normal operation of the ventilation system are located external to the control and pool rooms. Proper handling of airborne radioactive materials (in emergency situations) can be effected with a minimum of exposure to operating personnel.

5.7 Reactor Pool Water Systems

Applicability: This specification applies to the pool containing the reactor and to the cooling of the core by the pool water.

Objective: The objective is to ensure that coolant water shall be available to provide adequate cooling of the reactor core and adequate radiation shielding.

Specifications:

(1) The reactor core shall be cooled by natural convection water flow.

(2) All piping extending more than 5 ft below the surface of the pool shall have adequate provisions to prevent inadvertent siphoning of the pool.

(3) A pool level alarm shall be provided to indicate a loss of coolant if the pool level drops more than 2 ft below the normal level.

(4) The reactor shall not be operated with less than 15 ft of water above the top of the core.

Basis: This specification is based on thermal and hydraulic calculations which show that the TRIGA-30/20 LEU core can operate in a safe manner at power levels up to 2000 kW with natural convection flow of the coolant water. A comparison between operation of the TRIGA-30/20 LEU and standard TRIGA MARK III has shown them to be safe for the above power level. Thermal and hydraulic characteristics of mixed cores are essentially the same as those for TRIGA-30/20 LEU and standard cores.

In the event of accidental siphoning of pool water through system pipes, the pool water level will drop no more than 5 ft from the top of the pool.

Loss of coolant alarm after 2 ft of loss requires corrective action. This alarm is observed in the reactor control room, at the office, and at the campus police station.

5.8 Physical Security

The Licensee shall maintain in effect and fully implement all provisions of the NRC staff-approved physical security plan, including amendments and changes made pursuant to the authority of 10 CFR 50.54(p). The approved security plan consists of documents withheld from public disclosure pursuant to 10 CFR 2.70, collectively titled, “Washington State University, Pullman, Washington TRIGA Reactor Security Plan.”

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING CONVERSION ORDER TO CONVERT FROM

HIGH-ENRICHED TO LOW-ENRICHED URANIUM FUEL

FACILITY OPERATING LICENSE NO. R-76

WASHINGTON STATE UNIVERSITY TRIGA REACTOR

DOCKET NO. 50-027

1.0 INTRODUCTION

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.64 requires licensees of research and test reactors to convert from the use of high-enriched uranium (HEU) fuel to low-enriched uranium (LEU) fuel, unless specifically exempted. Washington State University (WSU or the licensee) has proposed to convert the fuel in the WSU Nuclear Radiation Center Reactor (NRCR) from HEU to LEU. In a letter dated August 15, 2007, as supplemented on December 14, 2007, and January 15, June 13, and August 4, 22, and 25, 2008, the licensee submitted its proposal for conversion requesting approval of the fuel conversion and of changes to its Technical Specifications (TSs). To support this action the licensee submitted a conversion Safety Analysis Report (SAR) on which the HEU to LEU conversion and the TS changes were based. After an initial review by the NRC, requests for additional information (RAIs) were issued to the licensee. The licensee then submitted, in letters dated June 13 and August 4, 22, and 25, 2008, an updated version of the SAR, responses to the RAIs, and the TSs that would be applicable after conversion. This Safety Evaluation Report provides the results of the NRC staff's evaluation of the licensee's conversion proposal. The evaluation was carried out according to the guidance found in NUREG-1537. [1]

On January 23, 2008, the NRC issued an Order to the licensee for the receipt and possession of the special nuclear material needed for the conversion. This was to allow the licensee to receive the LEU from the manufacturer in France before the certification of the shipping containers used to ship the fuel expired in June 2008.

2.0 EVALUATION

2.1 Summary of Reactor Facility Changes

The NRCR is a TRIGA-conversion reactor similar in design to many others operating in the U.S. and abroad. The reactor was originally designed for Material Testing Reactor (MTR)-type fuel assemblies. General Atomics (GA) developed a fuel system with fuel assemblies that can hold up to four TRIGA fuel elements each. These fuel assemblies were designed to replace the MTR plate-type fuel assemblies.

The reactor normally operates at a maximum thermal power level of 1 MW(t) although the TSs allow operation up to 1.3 MW(t) for calibration and testing purposes. The reactor uses natural

convection for cooling. It is presently fueled with highly-enriched (70% Uranium-235) TRIGA fuel lifetime improvement program (FLIP) fuel and low-enriched (less than 20% Uranium-235) TRIGA low density standard fuel elements (SFEs) (as compared to the conversion high density LEU fuel replacing the FLIP fuel). The HEU to LEU conversion only requires changes in the fuel type and core configuration; it does not require any changes to the remainder of the facility. The new conversion LEU fuel will have a nominal enrichment of 19.75% in a fuel element with the same geometry and with the same cladding as the FLIP fuel. Only the FLIP elements are being replaced by the conversion. The existing low-enriched SFEs will remain in use.

2.2 Comparison with Similar Facilities Already Converted

A similar TRIGA conversion reactor at Texas A&M University has converted using the same design LEU fuel elements proposed for the NRCR conversion. In addition, the TRIGA Mark F reactor at GA in San Diego was operating with a core partially made up of high density LEU conversion fuel when it was permanently shut down. There have been no performance issues in the use of this fuel in these reactors.

2.3 Fuel and Core Design

The physical size of the fuel assemblies or elements will not change in the conversion from HEU to LEU. The licensee plans to replace the HEU fuel elements and reuse the existing LEU SFEs. The outer diameter of the fuel meat will remain at 34.82 mm with a fuel meat length of 381 mm. There will be the same number of fuel elements (119) in the converted core as in the current core. Two of the fuel elements contain thermocouples and are referred to as instrumented fuel elements (IFEs). Table 1 (Table 3 in the conversion SAR) shows the fuel parameters for the existing FLIP and standard fuel and the new conversion LEU fuel.

The fuel meat composition in the new conversion LEU fuel is different from that in the FLIP or current SFE fuel. The FLIP fuel elements contain erbium poison at 1.48 weight percent (w/o). With the new conversion LEU fuel, the erbium poison will be lowered to 0.90 w/o. The TRIGA FLIP and existing TRIGA SFEs have a uranium content of 8.5 w/o (SFEs are referred to as 8.5/20 fuel) and the new conversion LEU fuel will have 30 w/o uranium content (the new conversion fuel is referred to as 30/20 fuel). The balance of the fuel is zirconium hydride (ZrH_x), where x is approximately 1.6 for the FLIP and new conversion 30/20 fuel and is approximately 1.7 for the previous 8.5/20 SFEs. The change in uranium content means there is less hydrogen in the new conversion 30/20 fuel. The reduced hydrogen content changes the value of the negative fuel temperature coefficient of reactivity, α (discussed below in Section 2.4.4). The amount of U-235 per fuel element will increase slightly from the FLIP fuel to the new conversion 30/20 LEU fuel.

Generic behavior of LEU TRIGA fuel with relatively high (30 w/o) uranium content has been previously approved for use in research and test reactors by the Nuclear Regulatory Commission (NUREG-1282).[2] The licensee submitted this application because it still needs to justify the use of the approved fuel in the NRCR.

The WSU control system consists of three control (shim) blades, a stainless steel regulating blade (servo blade) and a transient rod (water followed) (the control blades and transient rod will be referred to as rods). The transient rod will be replaced at the time of the conversion; all the control blades will remain. The replacement is not due to the conversion but rather because of normal maintenance that will be convenient to do at the time of conversion. The new transient rod will be of the same design as the existing one.

The staff has reviewed the proposed fuel and core design of the LEU reactor. The staff concludes that the conversion from HEU to LEU fuel will not impact the overall basic design of the core and its control. Therefore, the staff finds the fuel and core design acceptable.

Table 1. Description of TRIGA HEU and LEU Fuel Elements

Design Data	TRIGA FLIP HEU	TRIGA 30/20 LEU NEW CONVERSION	TRIGA 8.5/20 LEU SFE
Number of Fuel Elements at Full Load	51	51	68
Fuel Type	U-ZrH_x (FLIP)	U-ZrH_x (30/20)	U-ZrH_x (8.5/20)
Enrichment, %	70	19.75	19.75
Uranium Density	-	-	-
g/cm^3	0.5	2.14	0.5
wt-%	8.42	30	8.5
Number of Fuel Elements per Cluster	4	4	4
^{166}Er per Fuel Element, g	10.27	7.46	----------
^{167}Er per Fuel Element, g	7.09	5.15	----------
Erbium, wt-%	1.48	0.9	----------
Zirconium Rod Outer Diameter, mm	6.35	6.35	6.35
Fuel Meat Outer Diameter, mm	34.823	34.823	34.823
Fuel Meat Length, mm	381	381	381
Cladding Thickness, mm	0.508	0.508	0.508
Cladding Material	304 SS	304 SS	304 SS

2.4 Nuclear Design

This section of the SER discusses the nuclear design of the reactor and the impact that conversion will have. Some of the relevant neutronic parameters are summarized in Table 3 (extracted from Table 2 in the SAR) along with parameters that will be discussed in Section 2.5.

2.4.1 Calculational Methodology

The WSU reactor is a TRIGA conversion reactor and the analysis needed for conversion was performed by the supplier, GA. The computer codes utilized for the neutronic analysis were DIF3D (multigroup diffusion theory in three dimensions), MCNP5 (Monte Carlo), BURP/DIF3D (burnup) and GGC-5 (cross section generation). The first two of these computer codes are generic codes commonly used by many organizations for nuclear reactor analyses and are well-documented and benchmarked. The latter two are in-house codes that GA has developed and verified and successfully used for many years. Two different cores were modeled for the analysis. The first is the present core consisting of 51 FLIP HEU fuel elements and 68 partially burned 8.5/20 SFEs. The second is the proposed mixed LEU core which contains 51 new conversion 30/20 LEU fuel elements and the same 68 partially burned 8.5/20 SFEs. The first model was used to benchmark the analytical methods and the second was used to predict what will happen due to conversion. The current core of HEU FLIP fuel elements and LEU 8.5/20 SFEs is referred to as Core 34A (the mixed HEU core) and the proposed core of LEU 8.5/20

SFEs and new conversion 30/20 LEU fuel elements is referred to as Core 35A (the mixed LEU core).

The benchmarking of the MCNP5 core model was done with a comparison of measured to calculated reactivity of the mixed HEU core 34A when all of the control rods are fully withdrawn and the reactivity when all control rods are fully inserted. The total excess reactivity of the core when all of the control rods are withdrawn was calculated to be $6.31 and the measured value was $6.65, based on the measured control rod worths. This is a reasonable agreement. When the control rod worths were calculated for individual control rods, the total worth is $13.24[1] and the total worth based on individually measured control rod worths is $12.53. This difference of 6 percent is excellent agreement. The differences between the calculated and measured worths for individual control rods are higher but still reasonable taking into account both the difficulty in calculating individual rods and the error in the measurements. Because the licensee used documented codes that are well-accepted and have been validated against data for TRIGA reactors, including the one at WSU, the staff concludes that the calculational methodology used by the licensee is acceptable.

2.4.2 Core Parameters—Criticality and Control Rod Worths

The MCNP5 model predicts that initial criticality will occur with 47 of the new conversion 30/20 LEU fuel elements and 24 of the partially burned 8.5/20 LEU SFEs in the core. The core grid plate positions without fuel elements were replaced in the model with water and are nearest to the thermal column. All of the control rods are included in the analysis. There is no corresponding measured information for the present mixed HEU core since this core evolved over time. Because of that fact, no computer model of the mixed HEU core with a cold critical configuration was developed.

For the mixed HEU core, MCNP5 calculates k_{eff} = 1.05058 with all of the control rods withdrawn, which corresponds to a reactivity of $6.31 ($\beta_{eff}$ = 0.0076). For the mixed LEU core the values are k_{eff} = 1.05019 and ρ = $6.37 ($\beta_{eff}$ = 0.0075). The calculations with the control rods inserted resulted in values of k_{eff} = 0.95010 and ρ = -$6.91 for the mixed HEU core and k_{eff} = 0.94717 and ρ = -$7.44 for the mixed LEU core. The control rods are worth $13.24 for the mixed HEU core and would be essentially unchanged at $13.81 for the mixed LEU core. The calculated individual rod worths are slightly changed by conversion. TS 3.4, “Maximum Excess Reactivity,” limits the maximum excess reactivity of a cold, xenon-free core to $8.00 or less. The proposed mixed LEU core meets the requirement of TS 3.4.

Differential control rod values were not presented, only integral values were presented by the licensee. Differential rod worths are typically used to define limiting rate conditions for rod withdrawal reactivity accidents. However, in the case of a TRIGA type reactor, the reactivity insertion events as a result of normal pulsed behavior are bounding.

TS 3.2, “Reactivity Limitations,” states that the reactor shall not be operated unless the shutdown margin provided by the control rods is greater than $0.25 with:

A. The highest worth non-secured experiment in its most reactive state,

[1] This calculation, found in the responses to the RAIs, was done using a more accurate representation for the position of the control rods relative to what was used for the result presented in the SAR.

B. The highest worth control rod and the regulating rod (if not scrammable) fully withdrawn, and

C. The reactor in the cold condition without xenon.

The analysis used the control rod worth data from Table 9 for the mixed HEU core and Table 10 for the mixed LEU core in the SAR. Additional calculations were provided in the responses to the RAIs. The shutdown margin using the calculated values of the control rod worths for the mixed HEU core is -$1.61 and using the measured values of the control rod worths is -$0.70[2]. For the mixed LEU core the calculated value is -$0.91[3]. The calculated shutdown margin for the proposed mixed LEU core meets the requirements of TS 3.2.

The analysis of the shutdown margin described above includes the potential for a non-secured experiment to be suddenly removed. Technical Specification 3.10 (1), "Limitations on Experiments," allows non-secured experiments to have a reactivity worth up to $1.00.

2.4.3 Kinetics Parameters--β_{eff} and ℓ

The value of k_{eff} can be calculated with MCNP5 with and without delayed neutrons through a single computational switch. If the value of k_{eff} with delayed neutrons is k_t and the value of k_{eff} without delayed neutrons is k_p, then

$\beta_{eff} = 1- (k_p/k_t)$.

For the mixed HEU core the value of β_{eff} was calculated to be 0.0076 ± 0.0002 and for the mixed LEU core it was calculated to be 0.0075 ± 0.0002. The values are consistent with those found in other research reactors.

The prompt neutron lifetime, ℓ, was calculated with the 1/v absorber method, a standard technique where a small amount of boron is uniformly distributed throughout the reactor. The neutron lifetime is then correlated to the associated changes in core reactivity. For the mixed HEU core ℓ was calculated to be 30.7 μs and for the mixed LEU core ℓ was calculated to be 28.2 μs. Both of these values are consistent with those found in other research reactors.

2.4.4 Reactivity Coefficients

One of the significant safety features of TRIGA reactors is the large negative prompt temperature coefficient of reactivity. Even with prompt insertions of reactivity up to $5, the prompt feedback turns around power excursions before fuel damage due to overheating can occur. This feature allows TRIGA reactors to be pulsed. The key feature of the TRIGA fuels that makes this possible is that the fuel is uranium zirconium hydride ($UZrH_x$). The hydrogen in the fuel thermalizes the neutrons, and as the fuel heats up, the thermalization is reduced.

When the HEU is replaced with LEU, more uranium is needed and because the volume of a fuel element is fixed, less ZrH_x is used, so the effect of the hydrogen will be reduced which results in a smaller temperature coefficient. This is demonstrated in Table 2 which lists the calculated values of the temperature coefficient for the present mixed HEU core and for the proposed mixed LEU core under beginning-of-life (BOL) and end-of-life (EOL) conditions. Although the

[2] This value is based on recent measurements found in the responses to the RAIs.
[3] This value is based on revised calculations found in the responses to the RAIs.

feedback is reduced in going to the mixed LEU core (except at low temperature), it is still relatively strong.

Taking into account the change in reactivity coefficient, the amount of prompt reactivity insertion that can be allowed for the mixed LEU fuel might be smaller than it is for the mixed HEU fuel (see discussion in Section 2.4.6 below). According to the startup plan, the amount of reactivity that can be used for pulsing will be determined by initiating testing with $1.25 of prompt reactivity insertion. This will be increased in increments of $0.25 until the maximum fuel temperature, as set by the TSs, is approached. The safe maximum fuel temperature will set the cap on the amount of reactivity that can be used for pulsing.

Table 2. Values of the Calculated Negative Prompt Fuel Temperature Coefficient of Reactivity

Av. Core Temperature, °C	α (Δk/k/°C) HEU ($x10^{-5}$)	α (Δk/k/°C) LEU, BOL ($x10^{-5}$)	α (Δk/k/°C) LEU, EOL ($x10^{-5}$)
23 – 200	5.38	5.98	5.70
200 – 280	7.78	7.70	7.00
280 – 400	9.47	8.90	7.87
400 – 700	12.43	10.93	9.28
700 – 1000	15.10	12.69	10.64

The void coefficient of reactivity for TRIGA reactors is negative and for the mixed HEU fuel is calculated to be -0.08% Δk/k/% water void, and for the mixed LEU core is -0.135% Δk/k/% water void. If an evacuated experimental tube in the core were to be flooded, there would be a gain in the reactivity of the core. The SAR presents a calculation where a 205 cc dry experiment is placed in the core and it becomes flooded. The mixed HEU fuel will have an insertion of $0.10 reactivity and the mixed LEU fuel will experience an insertion of $0.18. Prompt critical occurs when the reactivity insertion passes $1.00, so a flooding of these experiments will not pose a risk to the reactor. It is also true that reactivity insertion events are bounded by what is allowed to be inserted for pulsed operation.

2.4.5 Effect of Burnup and Temperature

Curves of excess reactivity as a function of burnup are given in the SAR (Figure 30). After 1000 MWD of full power operation, and giving no credit for fuel shuffling to extend the life of the core, the mixed LEU core will have $0.40 of excess reactivity and the mixed HEU core would have had $0.67 of reactivity. After 1000 MWD there are no significant changes to the calculated reactor parameters, β_{eff} and ℓ. The value of β_{eff} is 0.0073 down from 0.0075, and the value of ℓ changes from 28.2 μs to 27.0 μs.

During operation the reactivity of the core drops due to the strong negative fuel temperature coefficient of reactivity. For the mixed HEU core the drop in going from critical to 1.0 MW(t) is calculated to be $1.73 whereas the measured value is $2.20. The difference is attributed to a larger gap in low power fuel elements resulting in higher average fuel temperatures. For the

mixed LEU core the drop at BOL is calculated to be $3.24 and at EOL is calculated to be lower due to the decrease in core average temperature with burnup. These values are typical.

2.4.6 Power Peaking and Pulsing Operations

The licensee provided calculations for steady state power peaking factors for the mixed HEU and the mixed LEU, BOL and EOL cores. The effect of conversion is predicted to decrease the power peaking from the present mixed HEU core. Therefore, conversion is not expected to have any negative impact on the power peaking.

The converted NRCR is expected to be able to operate for 13 years without fuel shuffling based on its anticipated operating schedule. During the life of the core it is conceivable that some new fuel may be introduced. There are three possible scenarios for this. Case 1 is the replacement of the partially burned 8.5/20 SFEs when they become more fully burned. The power factors in the region of the core where this fuel is located are very low. Therefore, introducing another partially burned 8.5/20 SFE from the irradiated fuel already on hand at WSU into the same core region will not lead to power peaking factors that exceed values considered in the analysis already done.

Case 2 is replacing an IFE that has been operated in the core with a fresh IFE because of thermocouple failure. In case 3 a fresh 30/20 LEU fuel element is introduced into the core. This could be to replace an 8.5/20 SFE that has reached the end of its life. These two cases could lead to higher power peaking factors especially if the fresh IFE is located near the transient rod water hole. This higher power peaking would be most likely during pulsing operation. The licensee would need to perform analysis of core changes but in general the licensee's proposed solution is "depending on the configuration of the core at the time that the fresh fuel is added, core locations 6C, 6E, or 2D should be considered for introducing any new fuel. These locations have very low peaking factors both at the beginning and the end of life." Analysis in the SAR shows that in any core location the power peaking from replacing a fuel rod that has been operated with a fresh fuel rod is not expected to exceed the power peaking expected at BOL which is bounding. The licensee's approach to replacement of fuel elements is acceptable.

The BLOOST computer code is a GA code that has been used for many years to calculate the performance of pulses in TRIGA reactors. For the WSU reactor, the BLOOST code overpredicts the energy generation for all pulses and overpredicts the maximum temperatures that are expected except for smaller reactivity insertions (Table 19 in the SAR). Conversion is expected to cause minimal change in response of the core to a pulse (see Tables 21 and 22 in the SAR). Initially the energy released by the mixed LEU core for a given reactivity addition as compared to the mixed HEU core is expected to be lower by about 10 percent, but at EOL the HEU mixed core and the LEU mixed core are expected to exhibit similar characteristics in power, energy release and maximum fuel temperatures. A startup plan is in place to assure that the Safety Limits are not exceeded by measuring the temperature at the two IFEs as a function of reactivity insertion (Section 2.4.4).

2.4.7 Conclusions

The calculations of neutronic steady state and dynamic parameters indicate that conversion of the reactor does not significantly change neutronic parameters and that conversion of the WSU NRCR should pose no significant problems or hazards. Therefore, the staff concludes that the changes in nuclear design of the core due to conversion are acceptable.

Table 3 Washington State University HEU – LEU Conversion Design Data

	Mixed HEU Core 34A	**Mixed LEU Core 35 A**
REACTOR PARAMETERS		
Reactor Power		
Licensed Power, MW	1.0	1.0
TS Max Power, MW	1.3	1.3
Max Fuel Temperature at 1 MW, °C	435 (b)	499 (c)
Max Fuel Temperature at 1.3 MW, °C	520 (b)	540 (c)
Cold Clean Excess Reactivity, Δk/kβ, $	7.17	6.94
Prompt Negative Temp. Coefficient of Reactivity -Δk/k-°C 23-1000°C	$0.54\text{-}1.51 \times 10^{-4}$	$0.60\text{-}1.27 \times 10^{-4}$
Coolant Void Coefficient, Δk/k per 1% void	-0.080%	-0.135%
Maximum Rod Power at 1 MW, kW/element	20.9	20.8
Average Rod Power at 1 MW, kW/element	8.4	8.4
Maximum Rod Power at 1.3 MW, kW/element	27.2	27.0
Average Rod Power at 1.3 MW, kW/element	10.9	10.9
Maximum Rod Power at DNB = 1.0, kW/element	51.7 (b)	45.8 (c)
DNB Ratio at Operating Power	2.47 (b)	2.20 (c)
Prompt Neutron Lifetime, μs	30.7	28.2
Effective Delayed Neutron Fraction	0.0076	0.0075
Shutdown Margin, Δk/kβ, $, with most reactive rod and Reg. Rod Stuck out	2.61	2.82
SAFETY PARAMETERS		
Limiting Safety System Setting, ˚C	500	500
LCO Max Power, MW	1.3	1.3
Minimum DNB ratio at 1.0 MW	2.47 (b)	2.20 (c)
Minimum DNB ratio at 1.3 MW	1.90 (b)	1.69 (c)
Calculated Maximum Positive Pulsed Reactivity Insertion to reach $\hat{T}$ =830°C, Δk/kβ, $ (a)	2.02	2.04 BOL 2.20 EOL
Peak Pulsed Fuel Temperature, °C	830	830

(a) Calculated maximum reactivity insertion for pulsing yields a conservative value, as shown by actual pulsing data.
(b) Pool water inlet temperature = 30°C
(c) Pool water inlet temperature = 50°C (maximum administrative limit)

2.5 Thermal-Hydraulic Design

The WSU NRCR conversion SAR presents the thermal-hydraulic analysis of steady-state operation in two parts. The first part presents the methodology and results for the mixed HEU core (core 34A). Operational data were used to benchmark the analytical results. The same computational technique was then used in the second part to analyze the mixed LEU core (core 35A). Results of the thermal-hydraulic analyses include fuel and coolant temperatures and the minimum departure from nucleate boiling ratio (MDNBR) at power of 1.0 and 1.3 MW(t). Some results are given in Table 3 above.

The NRCR fuel elements are cooled by natural convection. The driving force for the core flow is supplied by the column of water surrounding the core. A natural circulation flow rate is established to balance the driving head against the core entrance and exit pressure losses, and frictional, acceleration and hydrostatic head losses in the core flow channel.

The RELAP5 (version 3.2) code, a widely used and benchmarked system code for power and research reactors, was used for the thermal-hydraulic analysis. The steady-state analysis considered four different flow channels, each based on a four-rod cluster of fuel elements. The RELAP5 model examined a single fuel element and a representative flow area associated with that single fuel element. Each flow channel was assumed to experience the same driving head but the model ignored cross flow from one channel to another. Based on analysis and experiments in packed geometries similar to the WSU NRCR it is noted that cross flow tends to increase the heat flux at which critical heat flux (CHF) occurs in the limiting channel and thus neglecting cross flow is expected to result in conservative calculation of the CHF. The four channels represent the average channel (fuel rod with core average power), the maximum powered channel (maximum rod power) and two channels associated with the two IFE rods.

Given a inlet water temperature (30°C), reactor power (1.0 MW(t)), system pressure, local pressure loss coefficients and the axial power distribution for the bounding fuel elements, RELAP5 calculated the natural circulation flow rate, and along the axial length of the flow channel, the coolant temperature, wall heat flux, and the clad temperature. Geometric parameters for the flow channel and the fuel element are given in Tables 23 and 24 of the conversion SAR and the parameters are identical for both the mixed HEU and mixed LEU cores. The maximum powered element is determined by the DIF3D diffusion code and for both cores the hot rod (maximum powered element) is located in core position D4NE, which is not the location of the IFEs. The rod power peaking is represented by the rod power factor, defined as the power generation in a fuel rod (element) relative to the core averaged rod power generation. Two other power peaking factors are defined for the NRCR. They are the axial power factor (APF) and the intra rod peaking factor (Intra Rod). APF represents the axial peak-to-average power ratio within a fuel rod and Intra Rod represents the peak-to-average power in a radial plane within a fuel rod. The power peaking factors are given for the mixed HEU core 34A in Table 16 of the SAR and the corresponding values for the LEU mixed core 35A are given in Tables 17 and 18 for the BOL and EOL conditions, respectively.[4] The APF for the maximum power rod, the average rod, and the two IFEs are shown in Figures 31 and 32 for the mixed HEU core 34A and the mixed LEU core 35A (at BOL) respectively.

Results of the steady-state thermal-hydraulic analysis for the mixed HEU core 34A are summarized in Tables 25 and 26 of the conversion SAR. The RELAP5 code was also used to determine the MDNBR using the Bernath correlation, a CHF correlation historically used for TRIGA reactors. It is noted in the conversion SAR that the default CHF correlation in RELAP5 (Groeneveld 1986 Correlation) gave a higher CHF than the Bernath correlation. The departure from nucleate boiling (DNB) ratio (DNBR) is defined as the ratio of the CHF to the local heat flux. Thus a lower CHF or a higher local heat flux will lead to a lower DNBR. By accounting for peaking in the cladding wall heat flux, a minimum DNBR is determined. Given the maximum local heat flux occurs in the hot rod, the MDNBR is calculated as follows for core 34A:

> CHF (or DNB) is calculated to first occur when the maximum rod power reaches 51.7 kW(t).

[4] These tables are revised in the responses to RAIs.

The average power per element at the 1.0 MW(t) operating power = 1000 kW(t)/119 = 8.403 kW(t).

The maximum rod power factor (hot rod factor) for core 34A = 2.49.

The hot rod power at 1 MW(t) operating power = 8.403 kW(t) x 2.49 = 20.92 kW(t).

MDNBR = 51.7/20.92 = 2.47.

The corresponding MDNBR at a power level of 1.3 MW(t) = 2.47/1.3 = 1.90.

The thermal-hydraulic analysis is supplemented by the use of a finite-difference code TAC2D to calculate the steady-state fuel temperature. This GA code has been benchmarked and used for many TRIGA reactors. The code calculates temperatures in two-dimensional problems with radial and axial power distributions in the fuel given as input. The fuel rod model consists of the central zirconium rod, the fuel annulus, the fuel-to-clad gap, and the stainless steel clad. TAC2D requires a boundary condition given by an input from RELAP5, a clad surface temperature or a coolant temperature with the corresponding clad surface heat transfer coefficient. The gap between fuel and cladding was assumed to be filled with air. A cold gap of 0.2 mils was assumed throughout the core and some gap closure was expected to occur due to relative expansion of the fuel and cladding at normal operating temperatures. A summary of the calculated and measured fuel temperatures for the mixed HEU core 34A are shown in Table 27 of the conversion SAR. The calculated fuel temperatures at the sensing point of the two IFEs' thermocouples were about 6 to 7 percent higher than the measured fuel temperature at 1.0 MW(t) power level. The calculated maximum fuel temperature and the average fuel temperature at 1.0 MW(t) are 435 °C and 221 °C, respectively. SAR Section 4.5.7 discusses that based on the comparison of calculated versus measured reactivity loss (from cold critical to 1.0 MW(t)), the thermal calculation likely under predicts the core average temperature. This discrepancy is attributed to the average core gap of the lower powered fuel elements being slightly larger than the assumed 0.2 mils. A larger average gap will result in a higher core average temperature making the calculated and measured reactivity loss more consistent with each other. It is noted that for the mixed LEU core 35A a larger gap of approximately 2 mils was assumed for the fuel elements.

A similar thermal-hydraulic analysis as described above was conducted for the mixed LEU core 35A. The steady-state results for a 1.0 MW(t) operating power are summarized in Tables 28 and 29 of the conversion SAR. The RELAP5 calculations assumed more limiting power factors at the BOL conditions, namely a hot rod factor of 2.474 and an axial peaking factor of 1.286.

Results of the steady-state thermal-hydraulic analysis for the mixed LEU core 35A are summarized in Tables 28 and 29 of the conversion SAR. Given the maximum local heat flux occurs in the hot rod, the MDNBR is calculated as follows for core 35A:

CHF (or DNB) is calculated to first occur when the maximum rod power reaches 52 kW(t).

The average power per element at the 1.0 MW(t) operating power = 1000 kW(t)/119 = 8.403 kW(t).

The maximum rod power factor (hot rod factor) for core 35A = 2.474.

The hot rod power at 1 MW(t) operating power = 8.403 kW(t) x 2.474 = 20.79 kW(t).

MDNBR = 52/20.79 = 2.50.

The corresponding MDNBR at a power level of 1.3 MW = 2.5/1.3 = 1.92.

A summary of the calculated fuel temperatures by TAC2D for the mixed LEU core 35A are shown in Table 30 of the conversion SAR. The calculated fuel temperatures at the thermocouples of the two IFEs at 1.0 MW(t) power level are 427 °C and 440 °C. These calculated temperatures are for the bottom thermocouple (there are three thermocouples in each IFE) that tends to have the highest reading due to power tilting toward the lower half of the core from control blade insertion. The calculated maximum fuel temperature and the average fuel temperature at 1.0 MW(t) are 500 °C and 304 °C, respectively.

The thermal-hydraulic analysis was repeated for the mixed LEU core 35A with the pool water temperature at the administrative limit of 50 °C and the reactor power at the TS 3.1 limit of 1.3 MW(t) (the maximum permissible power for testing purposes while the nominal operating power is 1 MW(t)). At the combination of higher pool water inlet temperature and higher power the peak fuel temperature was increased by 40 °C (from 500 to 540 °C) while the peak clad temperature was increased by 22.4 °C (from 142.6 to 165 °C). The corresponding minimum DNBR (calculated by the Bernath correlation) was reduced from 2.5 to 1.69. Even at the bounding conditions of 50 °C pool inlet temperature and a reactor power of 1.3 MW(t), the thermal analysis shows that the facility can still operate safely.

The thermal-hydraulic analysis for the mixed LEU core 35A contains discussions of two effects that have impact on the MDNBR and the peak fuel temperature respectively.

Owing to its location adjacent to the transient rod, the maximum powered fuel rod (located in core position D4NE) is shown to have a flow area about 3.5 percent smaller than at other core positions. Basing on an analysis performed for a similar flow channel for the Texas A&M University HEU to LEU conversion [3], a 3.3 percent reduction in the calculated MDNBR is expected from the reduced flow area.

The effect of radial gap on peak fuel temperature is shown in Figure 36 of the conversion SAR, indicating a higher temperature for a wider manufactured gap because the gap acts to insulate the fuel meat. According to the conversion SAR, the manufactured radial gap between the fuel and cladding is limited to a maximum of 2 mils and the averaged radial gap is limited to less than 1.75 mils. From past operating experience, offset swelling of the fuel tends to close the gap once the core is put in operation. This implies a lower measured peak fuel temperature as the new core undergoes burnup. For the mixed LEU core 35A fuel temperature calculations, a gap of 1.75 mils was assumed for the average fuel element and a gap of 2 mils was assumed for the hot fuel element and the IFEs.

The staff concludes that the thermal-hydraulic analysis in the NRCR conversion SAR adequately demonstrates that the proposed conversion from an HEU to LEU core will result in no significant decrease in safety margins in regard to thermal-hydraulic conditions. The analyses were done with qualified calculational methods and conservative or justifiable assumptions. The applicability of the analytical methodology is demonstrated by comparing analytical results with measurements obtained from the mixed HEU core. In comparing the

HEU and LEU cores the thermal-hydraulic analyses have accounted for design differences in the fuel.

2.6 Accident Analysis

2.6.1 Failure of a Fuel Element Cladding in Air

To support the conversion from HEU to LEU fuel, the licensee performed a complete re-analysis of the Maximum Hypothetical Accident (MHA). The MHA for the WSU TRIGA reactor facility is an assumed cladding failure of one highly irradiated fuel element with no radioactive decay, followed by the instantaneous release of the noble gases and halogen fission products directly into the reactor pool room air. Boundary conditions and assumptions included using a conservative fuel element power density of 28 kW(t)/element with saturated inventories used for all released isotopes, and no credit for iodine absorption in the reactor pool water or dilution/filtration removal by the ventilation system. The occupational dose was calculated for an individual in the reactor pool room as well as the radiological exposure to the public, which included a direct release to the environment and the nearest resident, 600 meters to the southwest of the WSU reactor facility. Review of this methodology demonstrated the WSU TRIGA reactor facility analysis was consistent with the TRIGA MHA defined in NUREG-1537 and adequate to calculate occupational and public radiation doses.

Radioactivity releases from WSU TRIGA reactor facility operations can only occur if the fuel cladding is breached. A review of the licensee's analytically generated radionuclide inventory for the MHA was performed and the assumptions and boundary conditions used were consistent with accepted nuclear industry practices and representative of the proposed new LEU fuel. The use of a five minute occupational exposure is considered reasonable since evacuation drills conducted at the WSU facility have demonstrated the ability to evacuate personnel from the reactor pool room within that five minute timeframe.

Because there are no specific accident-related regulations for research reactors, the staff compared calculated dose values for accidents with related standards in 10 CFR Part 20 (the standards for protecting employees and the public against radiation). Amendments to 10 CFR Part 20 (20.1001 through 20.2402 and Appendices) became effective January 1, 1994. Among other things, these amendments changed the dose limits for occupationally exposed persons and members of the public, as well as the concentrations of radioactive material that are allowed in effluents released from licensed facilities. The licensee must follow the requirements of 10 CFR Part 20, as amended, for all aspects of operation regarding their facility. However, because the reactor was initially licensed before January 1, 1994, in conducting the accident evaluation, the staff used the dose limits in 10 CFR Part 20 that have been historically applied to accidents at this reactor (10 CFR 20.1 through 20.602 and Appendices, referred to as the "old" Part 20). See NUREG-1537, Chapter 13 for additional discussion of accident dose limits.

Review of the licensee's calculated results using the LEU source term for the MHA showed the thyroid and whole body occupational exposure was 2.77 rem thyroid and 0.21 rem whole body for an individual in the reactor pool room. This is below the occupational exposure limits of 30 rem thyroid and 5 rem whole body. The calculations for a representative exposure to the public, directly in the environment adjacent to the reactor facility showed exposure of 184 mrem thyroid and 0.8 mrem whole body, and exposure to the nearest resident showed exposure of 9.8 mrem thyroid and 0.04 mrem whole body. These exposures are below the public exposure limits of

3000 mrem thyroid and 500 mrem whole body. The staff notes that these doses are also within the limits of the current 10 CFR Part 20. Because these calculated occupational and public radiation exposures for the MHA are within the limits applied for licensing of this reactor, the staff finds the results of the MHA to be acceptable.

2.6.2 Loss-of-Coolant Accident (LOCA) Analysis

The LOCA analysis calculated what would happen under the limiting assumption that all water was removed from the pool. The time for removal is assumed to be 900 seconds based on the time it would take for the water to drain from an 8-inch beam tube after actuation of the low-level alarm for pool water. The licensee's acceptance criterion for standard fuel with H/Zr = 1.7 is given as 900 °C and for FLIP fuel with H/Zr = 1.6 as 940 °C. The 30/20 fuel has H/Zr in the range 1.57-1.65 and hence the 940 °C limit should apply. The staff has accepted if cladding temperature is greater than 500 °C, as could be expected during a LOCA with air cooling, no fuel damage is expected if the fuel temperature does not exceed 950 °C.

Calculations of the maximum fuel temperature during the LOCA vs fuel rod power density during operation were carried out assuming an infinite operating period. They show that the 8.5/20 SFE can be operated up to 22.3 kW(t)/element and the 30/20 LEU conversion fuel up to 23.5 kW(t)/element without exceeding the temperature limits. The maximum power density for this fuel is 20.8 kW(t)/element (in a 30/20 LEU conversion fuel element) for operation at 1.0 MW(t) so the temperature limits would be met in case of a LOCA. Allowing for a 4 percent level of uncertainty in reactor power the maximum power density in the fuel is 21.6 kW(t)/element which still meets the limits. If credit is also taken for the fact that the reactor only operates for 70 MW-hr/week or less (rather than use the assumption of an infinite operating period), then the bounding power densities are increased by 20 percent and even more margin exists. The licensee is allowed by the TSs to operate the reactor at up to 1.3 MW(t) for purposes of calibration and testing. Decay heat in a fuel element, which drives the temperature of a fuel element during a LOCA, is a function of operating history (a combination of power level and time at power). The amount of time the reactor is operated above 1.0 MW(t) for calibration and testing is limited. For example, for the period January 2, 2003, to August 20, 2008, the reactor was operated for 0.13 hours at a power level of 1.07 MW(t). Total operating hours during this period of time was 7554. Because the amount of time the reactor is operated above 1.0 MW(t) is limited, this operation does not significantly contribute to the total decay heat so that applying a maximum fuel element power density of 21.6 kW(t)/element to the LOCA evaluation is acceptable. These conclusions are supported by other studies performed for TRIGA fuels which show that, in general, as long as the operating power is less than 1.5 MW(t), the fuel cladding should not breach during a LOCA. Based on this analysis, the staff finds the results of the LOCA to be acceptable.

2.6.3 Accidental Pulsing from Full Power

An accidental pulsing accident from full power is hypothesized to be initiated by the ejection of an inserted transient rod (worth $3.19) or the unplanned removal of the maximum worth secured experiment ($2.00). The analysis utilized hand calculations and the BLOOST code. It showed that the resulting fuel temperatures in both accident sequences at BOL and EOL will be less than the Safety Limit of 1150 °C.

Furthermore, it is noted that these are low frequency events. The accidental pulsing of the transient rod requires the failure of the 1 kW(t) interlock that prevents air from being applied to the transient rod piston for reactor power above 1 kW(t) and failure of the reactor operator to

follow written procedures. The accidental removal of a secured experiment requires deliberate violation of written procedures as the operator unfastens the two-dollar experiment and pulls it out of the core quickly (assumed to happen in 0.3 s).

The staff concludes that there are sufficient design features and administrative restrictions in place to make accidental pulsing of the reactor unlikely and the safety limit will not be exceeded if it did happen.

2.6.4 Conclusions

The radiological consequences to the public and occupational workers at the WSU TRIGA reactor facility from postulated accidents analyzed for the proposed LEU-fueled reactor are expected to be similar to the radiological consequences calculated for the HEU-fueled reactor. The licensee has demonstrated that the consequences for the MHA for the conversion core are acceptable because they are within the limits of 10 CFR Part 20. Review of the calculations performed by the licensee and the assumptions made for the radiation source terms demonstrated that the inventory of radioactivity assumed and other boundary conditions used in the analysis are acceptable. As a result of this review, the staff has concluded that continued operation of the reactor poses no undue risk from a radiological standpoint to the public or the staff of the WSU reactor facility from the MHA.

The licensee identified fuel element power levels that could result in unacceptable fuel temperatures under LOCA conditions. The unacceptable temperatures could lead to cladding failure. The licensee showed that the peak power generated in a fuel element in the LEU core is less than the power levels that could lead to fuel failure. Therefore, the staff concludes that air cooling the reactor core after a LOCA is sufficient to prevent cladding failure.

The licensee did not identify any new reactivity addition accidents not previously analyzed for the HEU-fueled reactor. The design features and administrative restrictions that prevent accidental pulsing from occurring at full power and preclude damage if such pulsing occurs continue to exist for the LEU core. Therefore, risk to the health and safety of reactor staff and the public does not increase above that previously found acceptable for the HEU core from reactivity addition accidents.

2.7 Fuel Storage

In accordance with the existing technical specification (TS) 5.5(1), all reactor fuel assemblies shall be stored in a geometric array where the multiplication factor, K-eff, is less than 0.8 for all conditions of moderation and reflection. The licensee provided information on the storage of the new LEU conversion fuel in the fuel storage racks. For the five dry fuel storage racks, the licensee's analysis shows that the TS limit is satisfied with a considerable margin for both the normally dry condition (K-eff of 0.033) and under the assumption that the storage rack is flooded with water (K-eff of 0.376).

The licensee also performed a calculation of the multiplication factor assuming that either new LEU conversion fuel or FLIP HEU fuel is stored in the four reactor pool storage racks. The analysis shows that the TS limit is satisfied in both cases (K-eff of 0.649 for LEU fuel and K-eff of 0.635 for FLIP fuel). These values are within the multiplication factor limit of 0.8, so the fresh and spent fuel elements can be safely stored until used or removed.

The method used by the licensee for the analysis was the Monte Carlo code MCNP. MCNP is a state-of-the-art code frequently used for this type of analysis. The staff has reviewed the licensee's use of the code for all other aspects of its conversion analysis and found that it was knowledgeable in the application of the code. Hence, the staff has a high level of confidence in these results.

2.8 LEU Startup Plan

The initial startup is relatively straightforward and is based on the standard 1/M approach along with knowledge of how other TRIGA reactors have approached criticality. A full core is expected to have 51 fresh 30/20 LEU conversion fuel elements and 68 partially burned 8.5/20 SFEs. It is estimated that criticality will occur with 24 to 32 partially burned 8.5/20 SFEs and between 43 and 47 fresh 30/20 LEU conversion fuel elements.

Once the core has reached criticality, the reactivity worths of the control rods will be measured using the rod drop technique and the excess reactivity will likewise be measured using the period method. The control rods are expected to have worths between $0.40 and $3.20. As more fuel is added, the control rods are calibrated after sets of four fuel elements are added. After all of the fuel is loaded, the temperature coefficient of reactivity, α will be measured. A measurement is performed to assure that the reactor will be shut down by at least $0.25 with the most reactive control rod removed, and the control rods will be calibrated.

After the core loading has been completed, the reactor power will be calibrated using the calorimetric reactor power technique. In this technique, the indicated power is raised to 250 kW(t) and the rate of temperature rise of the pool water is measured. The power measurement channels are adjusted to this calibration and the power is raised to 750 kW(t) and the rate of temperature rise is determined again. If power measurement channels agree at 750 kW(t), the power is raised to 1 MW(t) and the acceptance criteria is that the power channel indicators agree to within 2 percent of the temperature rise measurements.

Full power of 1 MW(t) will be approached in steps where the power indicators, fuel temperatures, and control rod positions are recorded. The control rod positions will determine the excess reactivity of the core and that value will be compared to calculated values. This will proceed until 1 MW(t) has been achieved. As the power approaches 1 MW(t) the fuel expands and causes a permanent slight expansion of the fuel, so once the process is started, it will not stop until 1 MW(t) is achieved. After that, the process is repeated starting at 1 kW(t). As the fuel heats, there is a loss in core reactivity, so from 1 kW(t) to 1 MW(t) the acceptance criterion is for the core to lose between $2.24 and $2.85 of reactivity from the large negative fuel temperature coefficient of reactivity.

During the startup, the output of the fission chamber will be monitored as a function of power. An acceptance criterion is that a log-log plot of the fission chamber output current vs. reactor power will be nearly linear between 100 kW(t) and 1 MW(t).

One startup test is to increase the power to ensure a scram before the power exceeds 1.25 MW(t). That scram is one of the acceptance criteria.

Once the full power startup is completed, operation in pulsed mode will be tested. The focus is to find the maximum reactivity insertion while ensuring that the maximum fuel temperature is never exceeded.

Other aspects of the testing include looking at the effect of burnup. For all tests there is a contingency plan if the acceptance criterion/criteria are not met.

The licensee is to submit a start-up report to the NRC on the results of the start-up testing. The staff concludes that the licensee's testing program will provide verification of key LEU reactor functions and, therefore, is acceptable.

2.9 Proposed changes to License Conditions and Technical Specifications

For the HEU to LEU conversion, the licensee has proposed changes to the license conditions for special nuclear material possession limits and TSs.

2.9.1 Proposed Changes to License Conditions

License condition 2.B.2 is changed to reflect use of the special nuclear material needed for conversion. The license condition currently reads as follows:

(2) Pursuant to the Act and Title 10 CFR, Chapter I, Part 70, "Special Nuclear Materials", to receive, possess, and use up to a maximum of 25 kilograms of contained Uranium-235 at various enrichments and 32 grams plutonium contained in a plutonium-beryllium neutron source in connection with operation of the reactor. Without exceeding the foregoing maximum possession limits, the specific categories of maximum limits are as follows:

Maximum U-235	Maximum PU	% Enrichment	Exempt Status*
10.0 kg		< 20	Exempt 10 CFR 73.6(a)
15.0 kg		> 20	Exempt 10 CFR 73.6(b)
4.90 kg		> 20	Not Exempt
	32 grams		Exempt 10 CFR 73.6(c)

* Material is exempt provided that it meets the requirements for exemption pursuant to the cited provision of 10 CFR 73.

Based on the licensee's proposed possession limits, the license condition is amended to read as follows:

(2) Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material", to receive, possess, and use in connection with operation of the reactor:

a. up to 25 kilograms of contained uranium-235 enriched to less than 20 percent in the form of TRIGA reactor fuel.

b. up to 500 grams of contained uranium-235 enriched to any enrichment in the form of nuclear detectors and material for experimental research.

c. up to 32 grams of plutonium in the form of a plutonium-beryllium neutron source.

Up to 25 kilograms of contained uranium-235 of enrichment of less than 20 percent in the form of TRIGA reactor fuel replaces the existing possession limit. The reactor fuel consists of the new 30/20 LEU conversion fuel needed for conversion of the reactor and the existing 8.5/20 LEU SFEs. The licensee has provided justification for the proposed limit to allow possession of fuel on site and to allow for the acquisition of fuel in the future. After the reactor is converted, the licensee has a continuing need to receive, possess and use small amounts of HEU to allow continued operation of the reactor (e.g., nuclear chambers) and conduct of the experimental program (e.g., flux foils and fueled experiments). A new possession limit of up to 500 grams of contained uranium-235 of any enrichment in the form of nuclear detectors and material for experimental research is added to the license condition. The license condition is rewritten to agree with current formatting.

License condition 2.B.(4) was added to the license by order dated January 23, 2008, as part of the conversion process to allow the licensee to possess the LEU fuel needed for conversion prior to this order. The authority for possession of the LEU fuel has been moved to license condition 2.B.(2) as discussed above. License condition 2.B.(4) is revised to allow possession, but not use, of the existing HEU core until it is removed from the licensee's site. The revised license condition reads as follows:

2.B.(4) Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," to possess, but not use, up to 15 kilograms of contained uranium-235 at equal to or greater than 20 percent enrichment in the form of TRIGA fuel until the existing inventory of this fuel is removed from the facility.

The staff has reviewed the possession limits associated with conversion of the reactor and concludes that the limits are appropriate for the converted reactor.

License condition 2.C.(2), which incorporates the TSs into the license, is changed to incorporate the TS changes needed for conversion into the license as discussed below.

2.9.2 Proposed Changes to Technical Specifications

The licensee has proposed changes to the TSs to reflect the conversion to LEU fuel. In many of the TSs, reference to FLIP fuel is replaced by reference to the new 30/20 LEU conversion fuel. The discussion above shows that this direct fuel replacement is acceptable and therefore the changes to the TSs are acceptable to the staff. Some of the changes are only to the basis of the TS. While not discussed in detail below, changes to TS bases were reviewed by the staff and found consistent with the licensee's application for conversion. The staff concludes that the proposed changes to the TSs meet the requirements of 10 CFR 50.36.

Definitions

The licensee has proposed changes to definition 1.3, "Reactor Component." The definition "FLIP Fuel" is removed and reference to FLIP fuel in the definition of "Fuel Rod" and "Mixed Core" is removed to reflect the fact that the conversion will permanently end use of FLIP fuel in the reactor. A definition of "30/20 LEU Fuel" is added and reference to 30/20 fuel is added to

the definition of “Fuel Rod” and “Mixed Core.” The definition of TRIGA 30/20 LEU Fuel reads as follows:

> 30/20 LEU Fuel: 30/20 LEU fuel is TRIGA fuel that contains a nominal 30 weight percent of uranium with a nominal ^{235}U enrichment of 19.75% and erbium, a burnable poison.

The definition of Fuel Rod has been changed to read as follows:

> Fuel Rod: A fuel rod is a single TRIGA-type fuel rod of either Standard or 30/20 LEU fuel.

The definition of Mixed Core has been changed to read as follows:

> Mixed Core: A mixed core is a core arrangement containing Standard and 30/20 LEU-type fuels with at least 51 30/20 LEU fuel rods located in the central positions in the core.

New definitions have been added to take into account the removal of the HEU FLIP fuel and the presence of the new 30/20 LEU conversion fuel and are therefore acceptable to the staff.

Safety Limit and Limiting Safety System Settings (LSSS)

The properties and performance of the TRIGA higher uranium weight percent LEU fuel, including the new 30/20 LEU conversion fuel proposed for the NRCR, have been evaluated by the NRC in NUREG-1282 [2] and approved for use with the provision that case-by-case analyses discuss individual reactor operating conditions in applications for authorization to use them. The NRCR LEU core adopts the fuel safety limit of 1150 °C in TS 2.1, “Safety Limit – Fuel Element Temperature,” as established in NUREG-1282 for the new 30/20 LEU conversion fuel. The safety limit for FLIP fuel is removed as FLIP fuel will not be used again in the reactor. The previous Safety Limit for the 8.5/20 SFE fuel remains at 1000 °C. TS 2.1 (2), the safety limit for the new 30/20 LEU conversion fuel reads as follows:

> (2) The maximum temperature in a 30/20 LEU-type TRIGA fuel rod shall not exceed 1150 °C under any condition of operation.

The licensee has made changes to the bases of TS 2.1 to reflect the conversion from FLIP to LEU fuel. The changes to TS 2.1 are acceptable to the staff because they reflect the removal of the HEU FLIP fuel and the presence of the new 30/20 LEU conversion fuel in the reactor.

The LSSS given in TS 2.2, “Limiting Safety System Settings,” does not change for the LEU fueled reactor and remains at 500 °C as measured in an IFE. Changes are proposed in the TS to reflect the replacement of FLIP fuel with new 30/20 LEU conversion fuel and to reflect the limitations on the position of the IFE in the core as follows:

> The limiting safety system settings shall be 500˚C as measured in an instrumented fuel rod in selected locations in the central region of the core. For a mixed core, the instrumented rod shall be located in one of the following positions in the region of the core containing the 30/20 LEU-type fuel rods: D2NE, D2SE, C3 (except for C3NE), D3, E3 (except for E3SE), C4, E4NE, E4NW, C5 (except for C5SW), D5SE, E5NE, E5NW, C6NW, or D6.

The IFE is now required to be in the region of the core containing the new 30/20 LEU conversion fuel, however, additional constraints are placed on the location in order to assure that the hottest location not only is protected against the safety limit but also against DNB. In addition, the constraints assure that a temperature trip would not take place before the high-power-level trip at 1.2 MW(t) to allow the power level trips to actuate first (1.2 MW(t) is the set point the licensee uses during operation. The TS limit is 125 percent of full licensed power or 1.25 MW(t)). The TS Basis contains addition details about the proposed LSSS. In the proposed LSSS uncertainties are taken into account, while still leaving a safety margin of 350 °C, which is identical to the HEU core. Because the LSSS of 500 °C protects the safety limit as proposed by the licensee, it is acceptable to the staff.

Limiting Conditions for Operation (LCOs)

The licensee has proposed a change to TS 3.1, "Steady-State Operation." The propose TS reads as follows:

> The reactor power level shall not exceed 1.3 MW under any condition of operation. The normal steady-state operating power level of the reactor shall be 1.0 MW. However, for purposes of testing and calibration, the reactor may be operated at higher power levels not to exceed 1.3 MW during the testing period.

The change restricts operation over 1.0 MW(t) to testing and calibration purposes only. The purpose is to help ensure that decay heat in the reactor will be limited such that a LOCA as discussed above will not result in fuel cladding failure. The proposed wording of TS 3.1 is identical to that which existed prior to the issuance of Amendment No. 14 in January 1994. Because the changes to this TS help ensure that the fuel cladding failure will not occur during a LOCA, it is acceptable to the staff.

The LCO for pulse mode operation leaves unchanged the limit on reactivity so that the peak fuel temperature remains below 830 °C. The reason the temperature limit is not changed is the fact that the hydrogen to zirconium ratio is the same for the FLIP and new 30/20 LEU conversion fuel and this ratio is key to fuel behavior under pulsed conditions. The basis of the TS is modified to reflect the conversion to LEU fuel.

In TS 3.4, "Maximum Excess Reactivity," the specification is given in percent delta k/k (primary value) and in dollars (secondary value). The licensee has requested that the specification be given only in percent delta k/k. The conversion of values in percent delta k/k to dollars is dependent on the value of the effective delayed neutron fraction which can depend on core design and can change during core life. Giving the specification in only percent delta k/k is acceptable to the staff.

TS 3.5, "Core Configuration Limitation" specifies how fuel is placed in the core. In TS 3.5 (1), reference to FLIP is replaced with the new 30/20 LEU conversion fuel in the Applicability, Specification and Bases of the TS. In TS 3.5 (2), the use of the peak-to-measured-fuel temperature ratio is replaced by the method used to determine the acceptable locations of the IFE as discussed above in the section on LSSS. The use of the peak-to-measured-fuel temperature ratio no longer has any meaning in the determination of the placement of the IFE in the core. The TS reads as follows:

(1) The 30/20-fueled region in a mixed core shall contain at least 51 30/20 fuel rods in a contiguous block of fuel in the central region of the reactor core. Water holes in the 30/20 region shall be limited to nonadjacent single-rod holes.

(2) Each new mixed core configuration shall be evaluated to determine the allowed locations for the IFE (Reference: Response to RAI Question Number 40).

Because the changes to this TS reflect the conversion to LEU fuel, the changes are acceptable to the staff.

Design Features

The proposed changes to TS 5.1, "Reactor Fuel," and TS 5.2, "Reactor Core," account for the replacement of FLIP fuel with 30/20 LEU conversion fuel. The changes are consistent with the information on conversion provided in the conversion SAR. In TS 5.1(1), design requirements for FLIP fuel are replaced with design requirements for 30/20 LEU conversion fuel. The proposed TS 5.1(1) reads as follows:

(1) 30/20 LEU Fuel – The individual unirradiated 30/20 fuel elements shall have the following characteristics:

- uranium content: maximum of 30 wt%; enriched to maximum of 19.95% with nominal enrichment of 19.75% U-235
- hydrogen-to-zirconium ratio (in the ZrH_x): nominal 1.6 H atoms to 1.0 Zr atoms with a maximum H to Zr ratio of 1.65
- natural erbium content (homogeneously distributed): nominal 0.90 wt%
- cladding: 304 stainless steel, nominal 0.020 in. thick

The basis of the TS is revised to reflect the conversion to LEU fuel. Because the TS reflects the conversion to LEU fuel it is acceptable to the staff.

The proposed TS 5.2(2) and (3) read as follows:

(2) The TRIGA core assembly may be composed of Standard fuel, 30/20 LEU fuel, or a combination thereof (mixed cores) provided that the 30/20 LEU fuel region contains at least 51 30/20 LEU fuel rods located in a contiguous block in the central region of the core.

(3) The reactor fueled with a mixture of fuel types shall not be operated with a core lattice position vacant in the 30/20 LEU fuel region. Water holes in the 30/20 LEU region shall be limited to single-rod holes. Vacant lattice positions in the core fuel region shall be occupied with fixtures that will prevent the installation of a fuel bundle.

The basis of the TS is revised to reflect the conversion to LEU fuel. Because the TS reflects the conversion to LEU fuel it is acceptable to the staff.

The licensee has proposed changes to the basis of TS 5.7, “Reactor Pool Water Systems,” to update the basis for the reactor being cooled by natural convection water flow. The changes to the basis are consistent with the thermal-hydraulic discussion above and are therefore acceptable to the staff.

2.9.3 Conclusions

The staff has reviewed all of the proposed changes to the license and TSs and finds them to be acceptable. The staff concludes that these changes to the license and TSs are needed for the conversion of the reactor to LEU fuel. The licensee has justified the technical bases for these changes as discussed above. The safety limit continues to protect the integrity of the primary barrier (fuel cladding) that protects against the uncontrolled release of radioactivity in accordance with 10 CFR 50.36(d)(1). Conservative temperature limits as measured in an IFE were used in the development of the LSSS and pulsing LCOs and these limits continue unchanged in the TSs.

3.0 ENVIRONMENTAL CONSIDERATION

In accordance with 10 CFR 51.10(d) of the regulations, an order is not subject to Section 102 of the National Environmental Policy Act. The NRC staff notes, however, that even if these changes were not being imposed by an order, the changes would not require an environmental impact statement or environmental assessment.

The changes involve use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes in inspection and surveillance requirements. The NRC staff has determined that the changes involve no significant hazards consideration, no significant increase in the amounts, and no significant change in the types, of any effluents that may be released off site, and no significant increase in individual or cumulative occupational radiation exposure.

Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment is required.

4.0 CONCLUSIONS

The NRC staff has reviewed and evaluated the operational and safety factors affected by the use of LEU fuel in place of HEU fuel in the NRCR. The staff has concluded, on the basis of the considerations discussed above that (1) the proposal by the licensee for conversion of the reactor to LEU fuel is consistent with and in furtherance of the requirements of 10 CFR 50.64; (2) the conversion, as proposed, does not involve a significant hazards consideration because the amendment does not involve a significant increase in the probability or consequences of accidents previously evaluated, create the possibility of a new kind of accident or a different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety; (3) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed activities; and (4) such activities will be conducted in compliance with the Commission's regulations and the issuance of this Order will not be inimical to the common defense and security or the health and safety of the public. Accordingly, it is

concluded that an enforcement order as described above should be issued pursuant to 10 CFR 50.64(c)(3).

5.0 REFERENCES

1. "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Standard Review Plan and Acceptance Criteria," NUREG 1537, Part 2, U.S. Nuclear Regulatory Commission, February 1996.

2. "Safety Evaluation Report on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors," NUREG-1282, U.S. Nuclear Regulatory Commission, August 1987.

3. "Issuance of Order Modifying License No. R-83 to Convert from High- to Low-Enriched Uranium Fuel (Amendment No. 17) – Texas A&M University Nuclear Science Center TRIGA Research Reactor (TAC No. MC(9449)," U.S. Nuclear Regulatory Commission, September 1, 2006.

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Dated: September 4, 2008